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Title: Calculation of Eigenfunction Fluxes in Nuclear Systems

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Los Alamos
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Calculation of Eigenfunction Fluxes in Nuclear Systems

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Nuclear Design and Risk Analysis (D-5)
Los Alamos National Laboratory

Purdue University
School of Nuclear Engineering

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Outline

Problem: Eigenfunction fluxes converge poorly

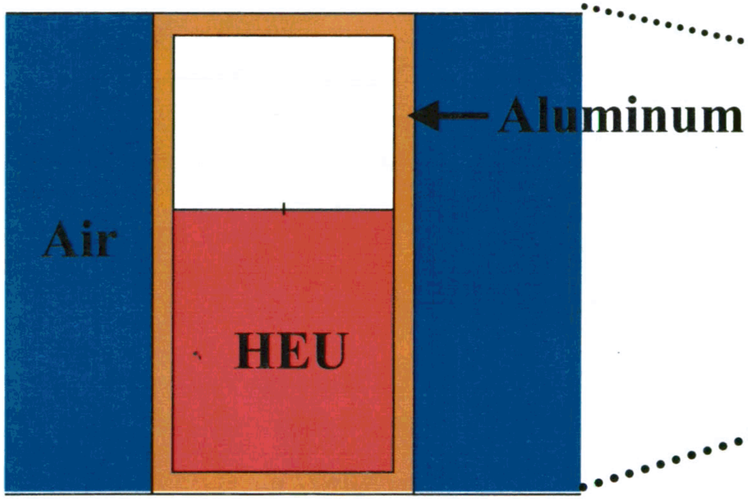
- 7-can lattice
- 2-sphere Godiva
- 11-segment fuel pin

Solution: Impose a fission distribution

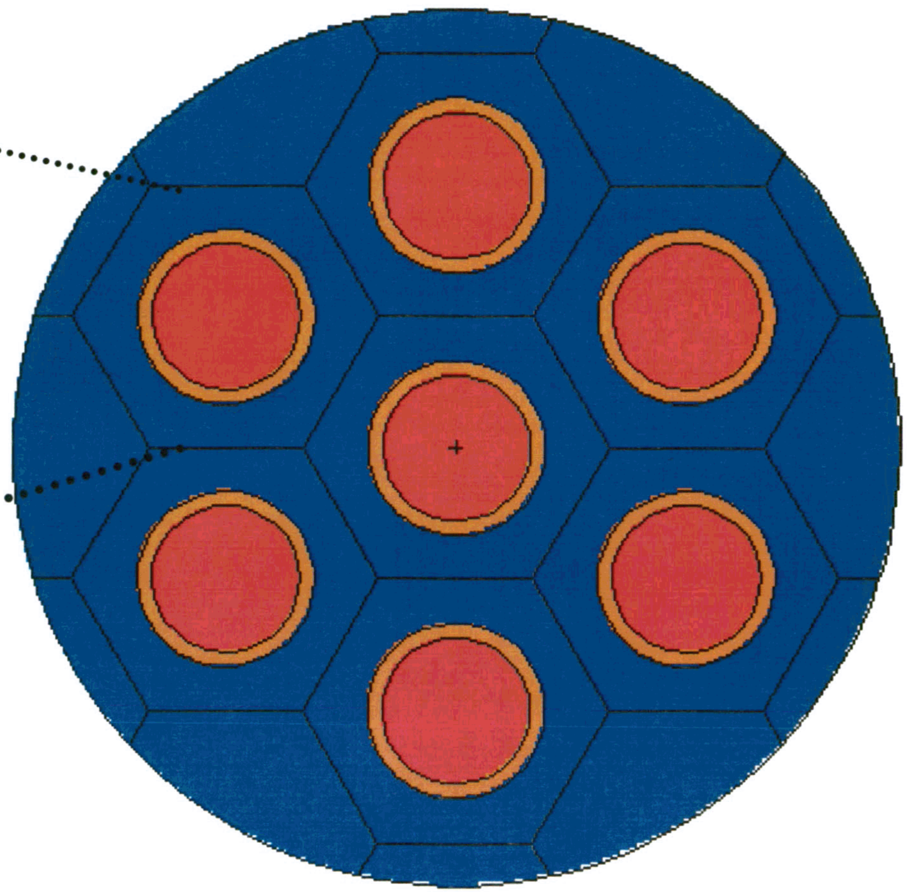
- What? How?
- Proposed method is preliminary

Goal: Unbiased, Quick, Simple, Available

7-Can Lattice Geometry



~5% Enriched



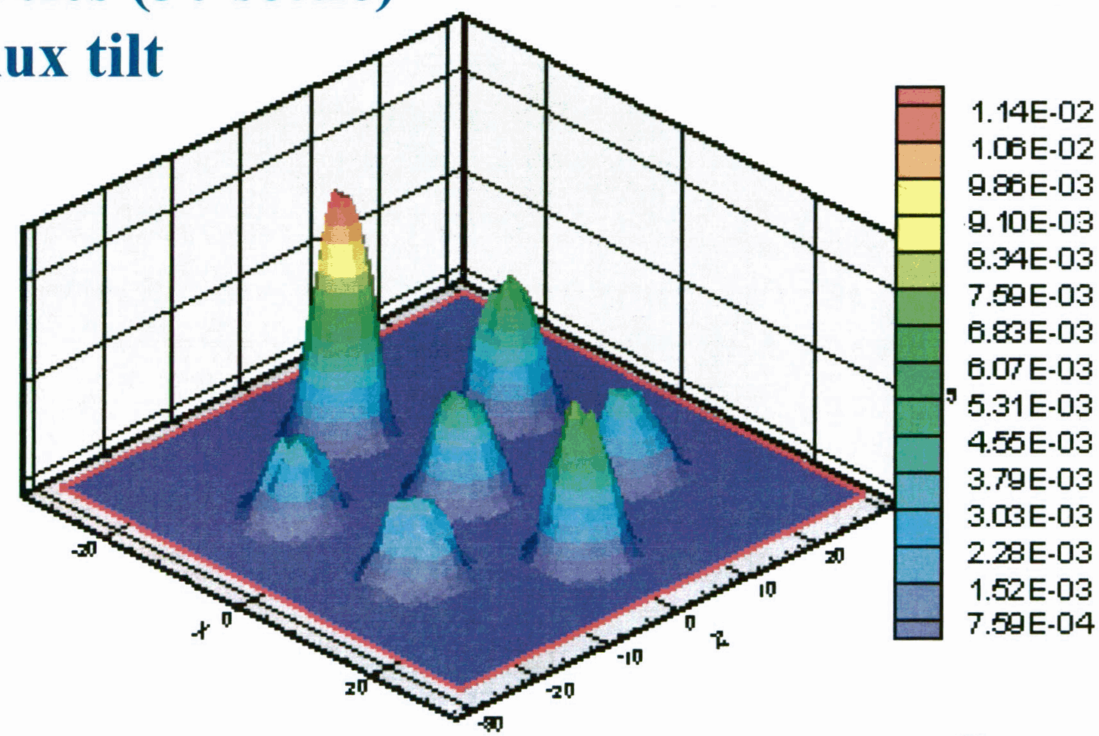
7-Can Lattice Description

- Seven identical cylindrical barrels
  - Fissionable material : 12 cm height, 5 cm radius
    - 8.4 g/cc, 0.118at%^{235}U, 2.268at%^{238}U, 35.72at%^{16}O, 61.894at%^{1}H
  - Void region atop fissionable material : 8 cm height
  - Aluminum "clad" : 1 cm thick

- Barrels separated by air : 18 cm center-to-center

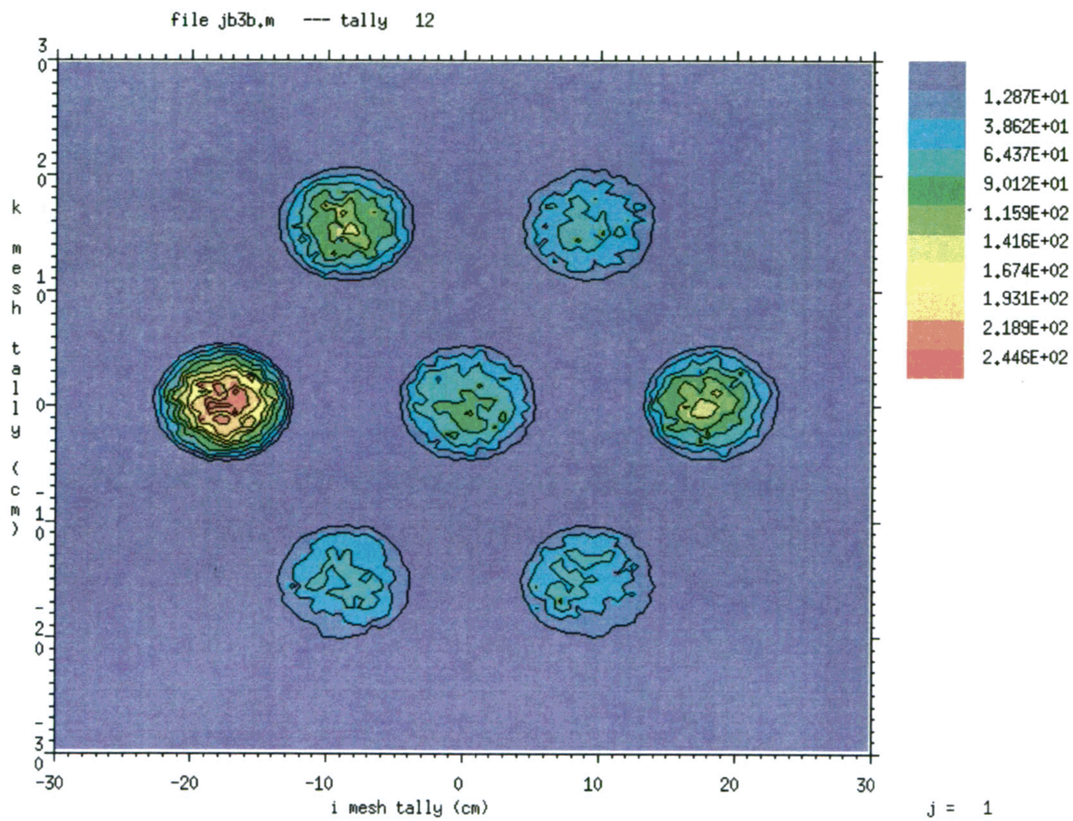
Mesh Tally of Results

- Standard method
- 1,000 particles/cycle
- 100 active cycles (30 settle)
- Factor 3-4 flux tilt



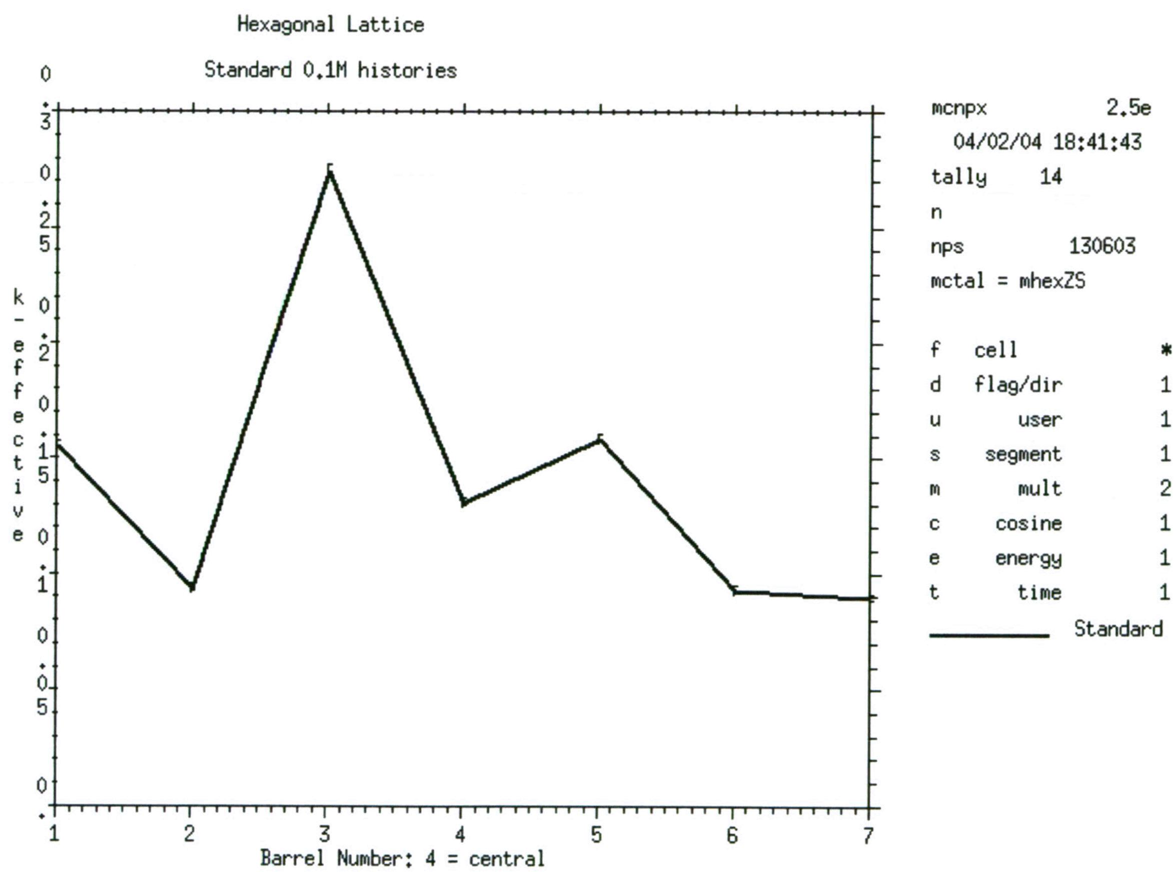
7-Can Lattice

Standard MC : 0.1M histories



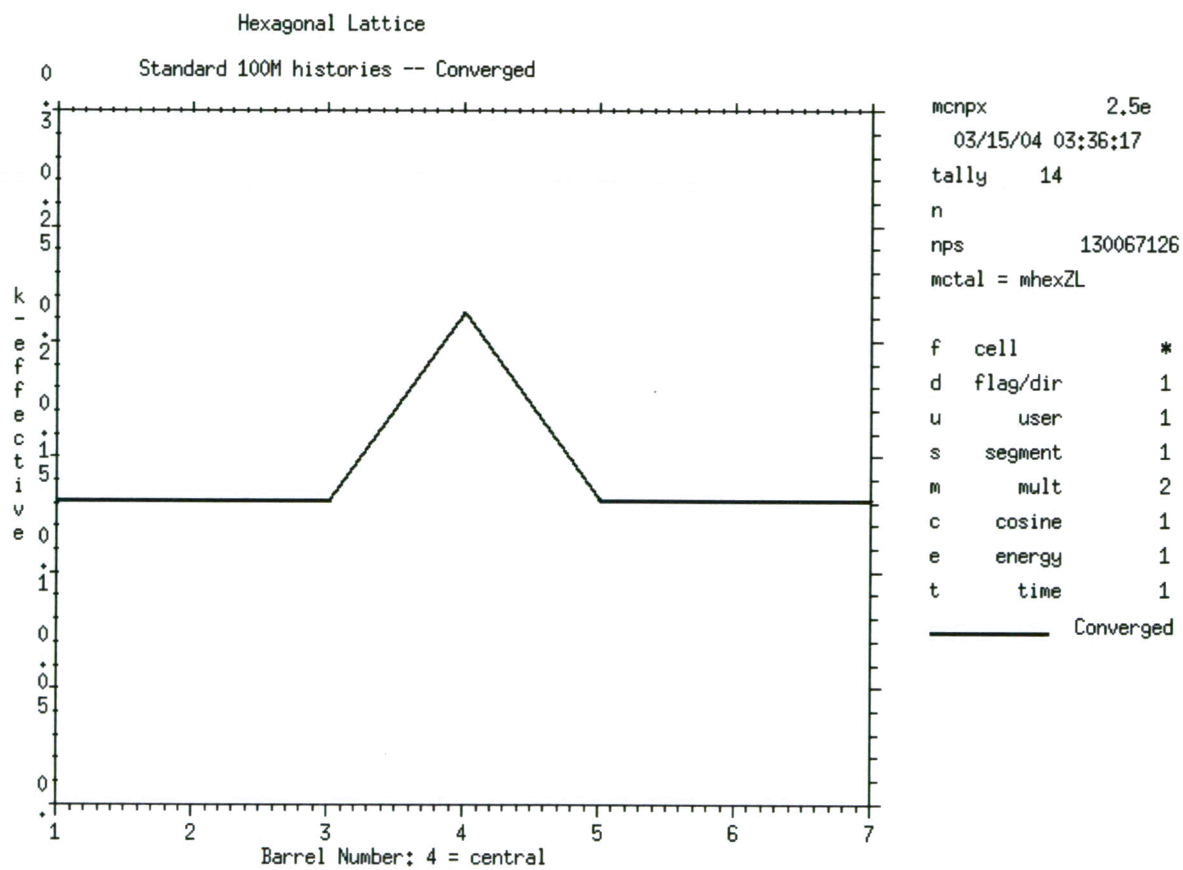
7-Can Lattice

Standard MC : 0.1M histories



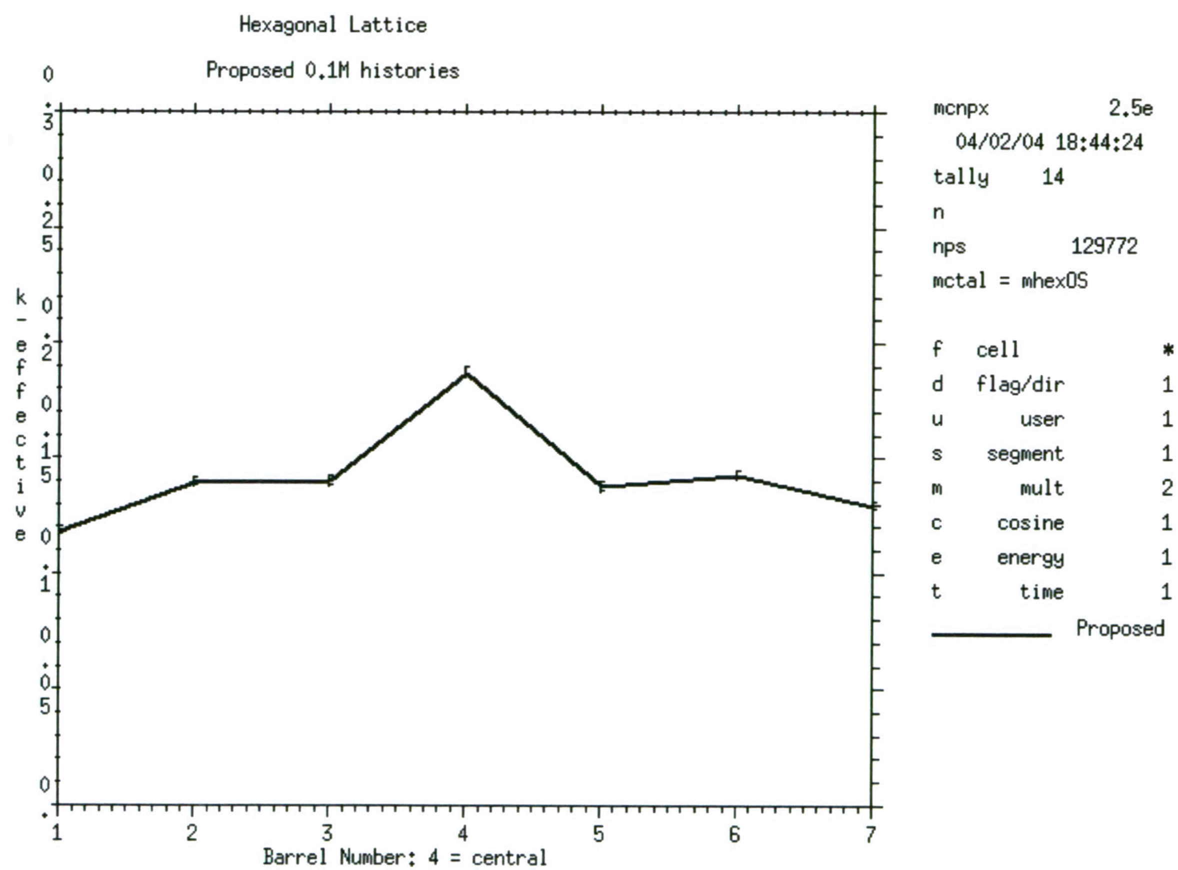
7-Can Lattice

Converged : Standard MC : 100M histories



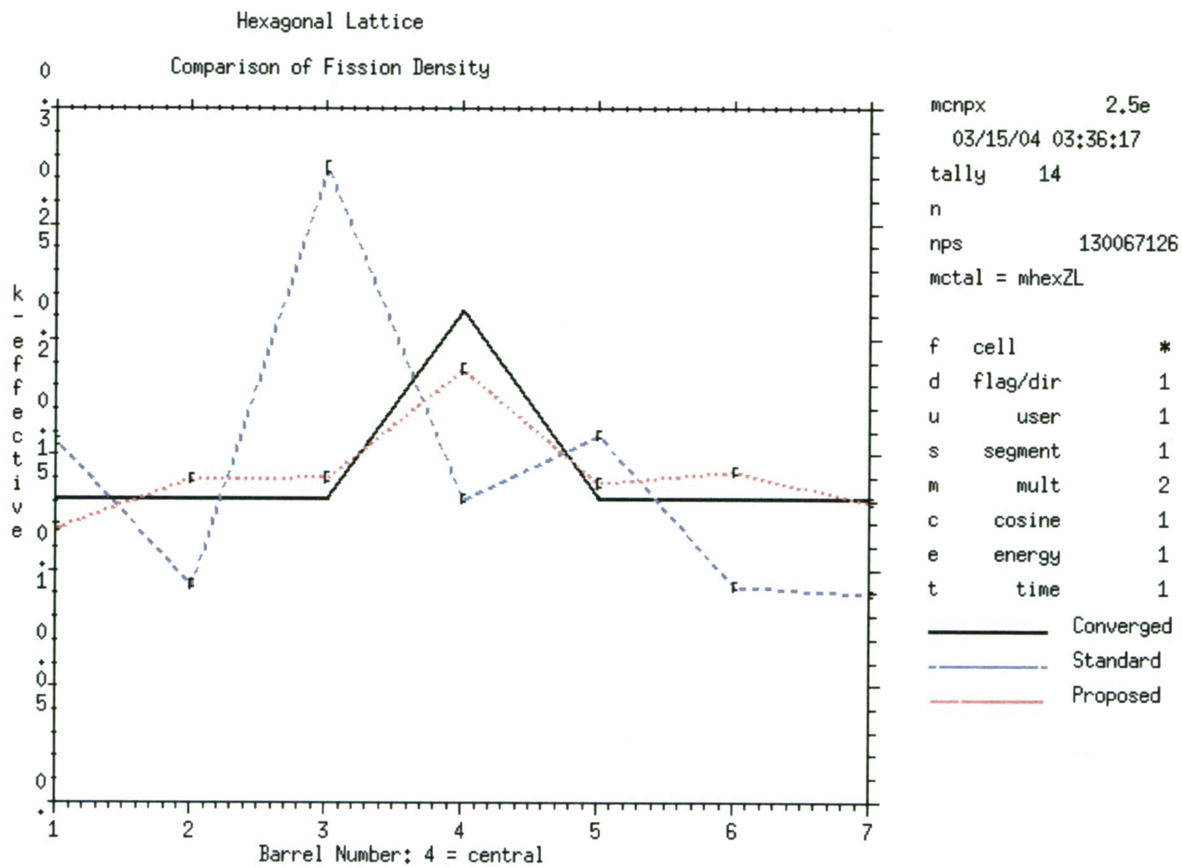
7-Can Lattice

Proposed : 0.1M histories

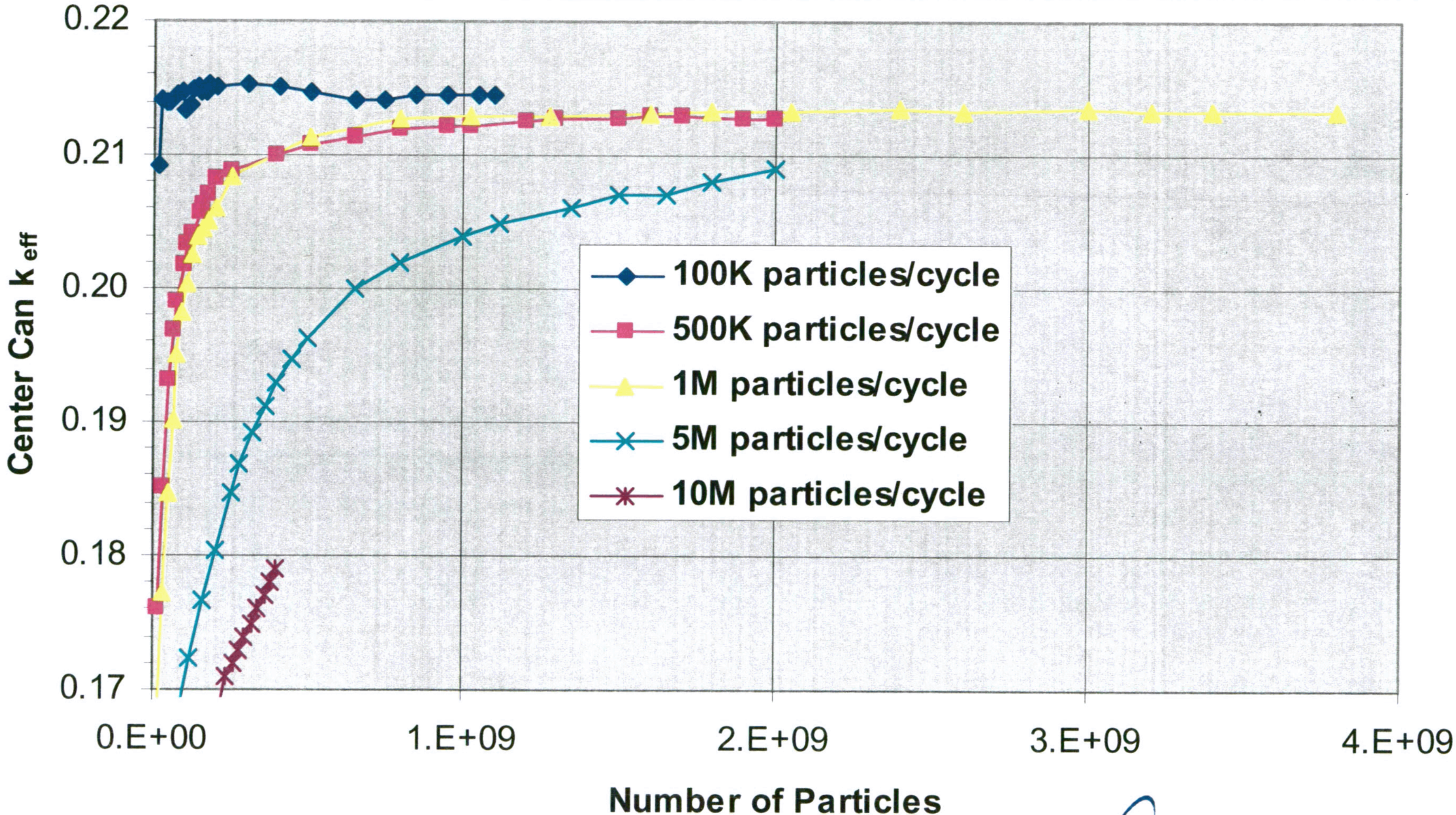


7-Can Lattice

Results



Convergence Results



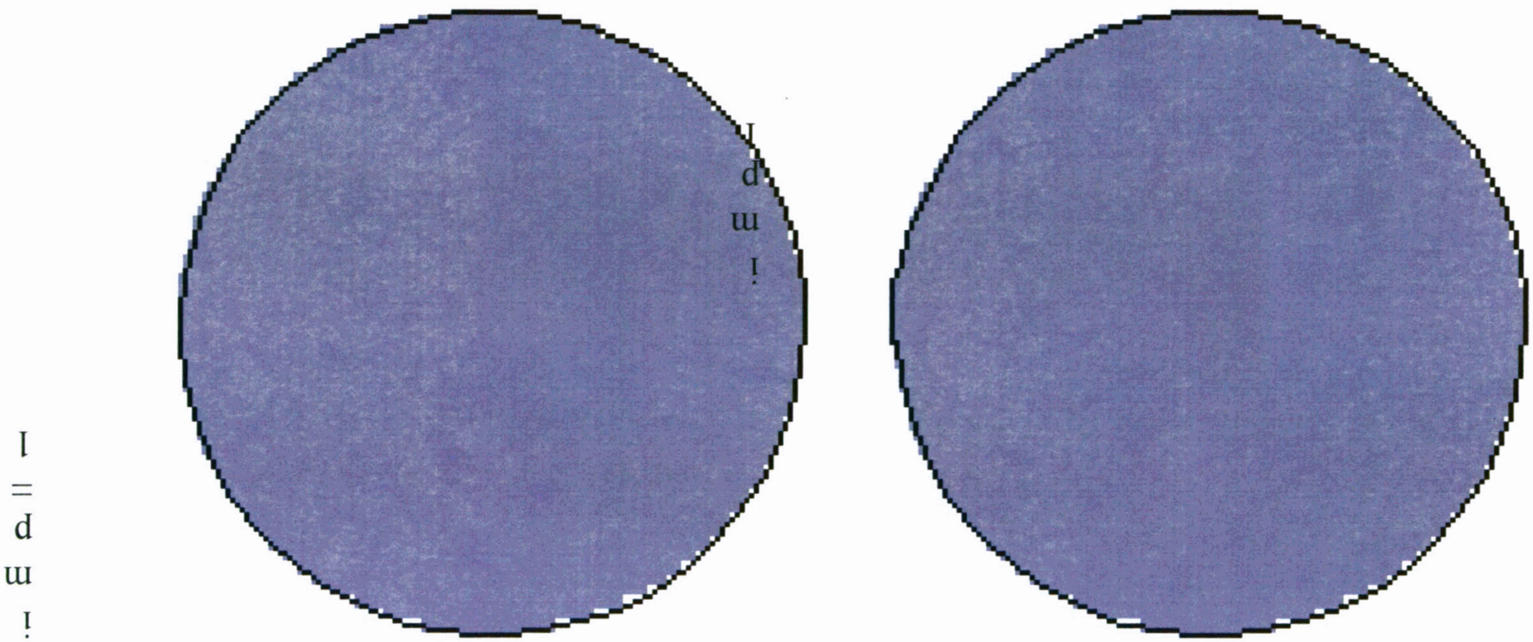
Convergence in Standard MC

- Hexagonal Lattice requires:

  $\geq$ 2 Billion NPS to converge
- NPS $\leftrightarrow$ Time

  This translates to days of computing
- Proposed method requires ~ 2 minutes

Two-Sphere Godiva Geometry



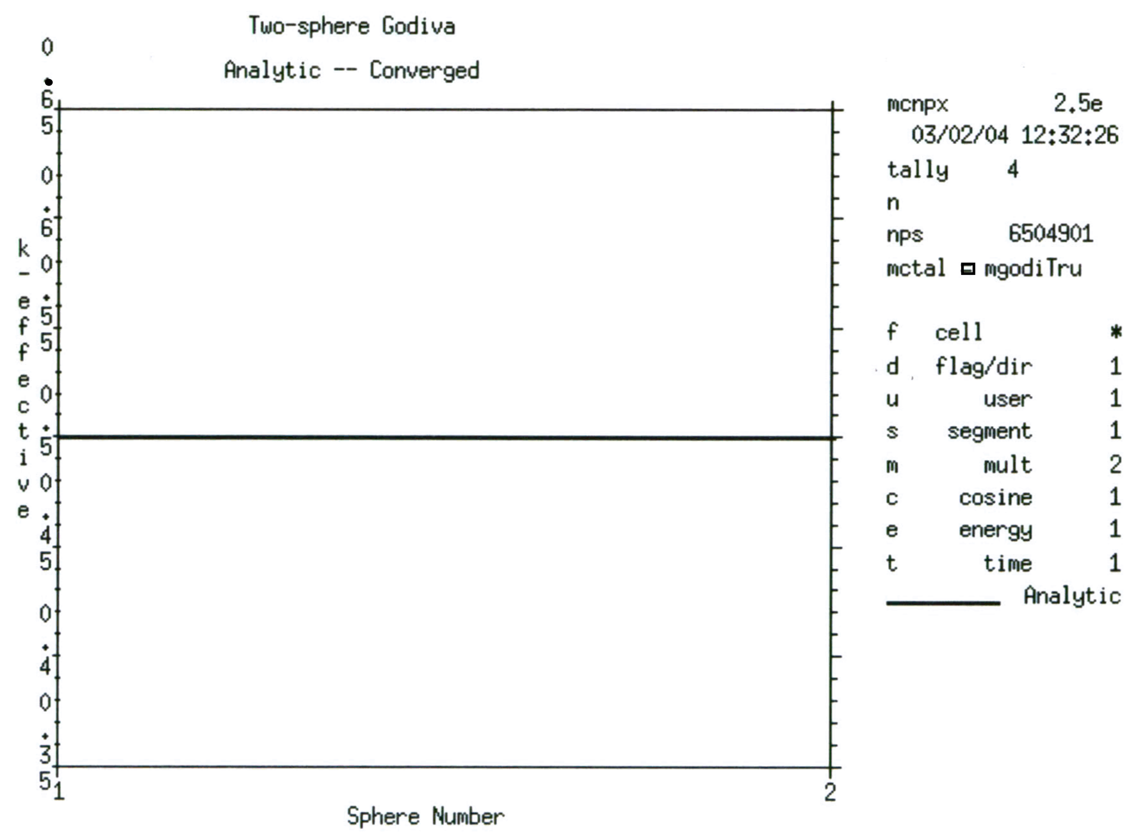
Two-Sphere Godiva Description

- Spheres
  - Radius = 8.741 cm
  - Composition : 94.73 wt% ^{235}U and 5.27 wt% ^{238}U

- Black Absorber separates spheres (zero importance region)
- 20-cm center-to-center separation

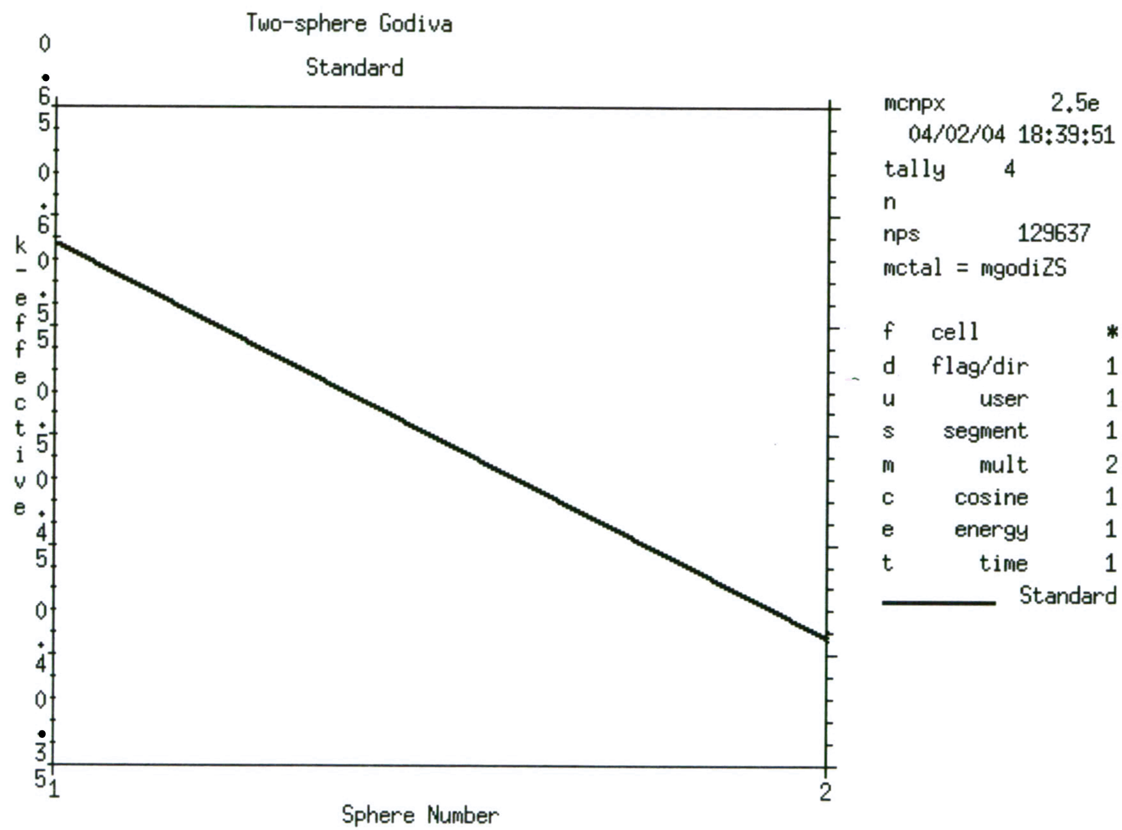
Two-Sphere Godiva

Analytic Solution



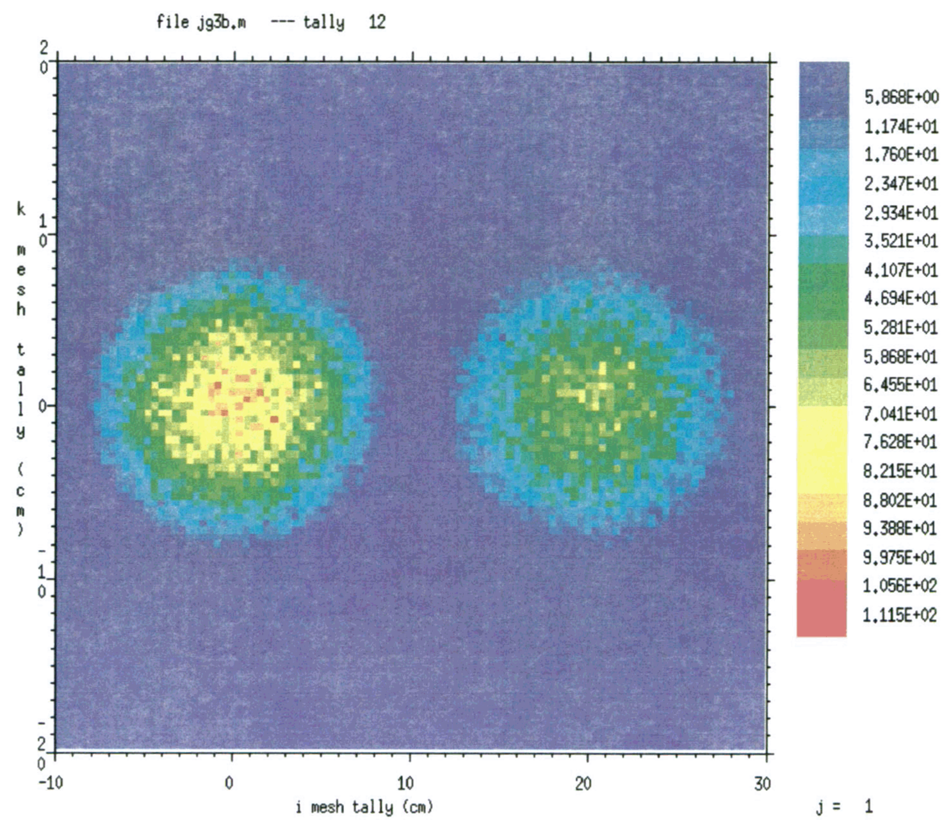
Two-Sphere Godiva

Standard MC : 0.1M histories



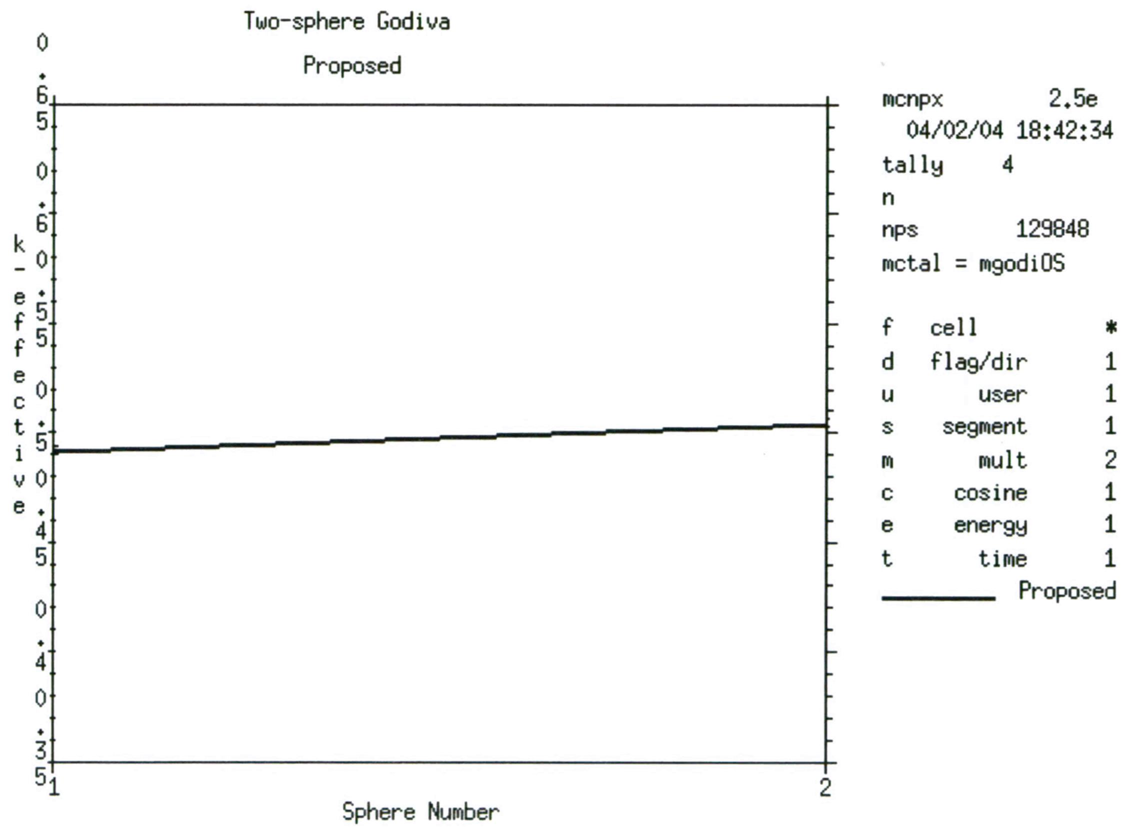
Two-Sphere Godiva

Standard MC : 0.1M histories

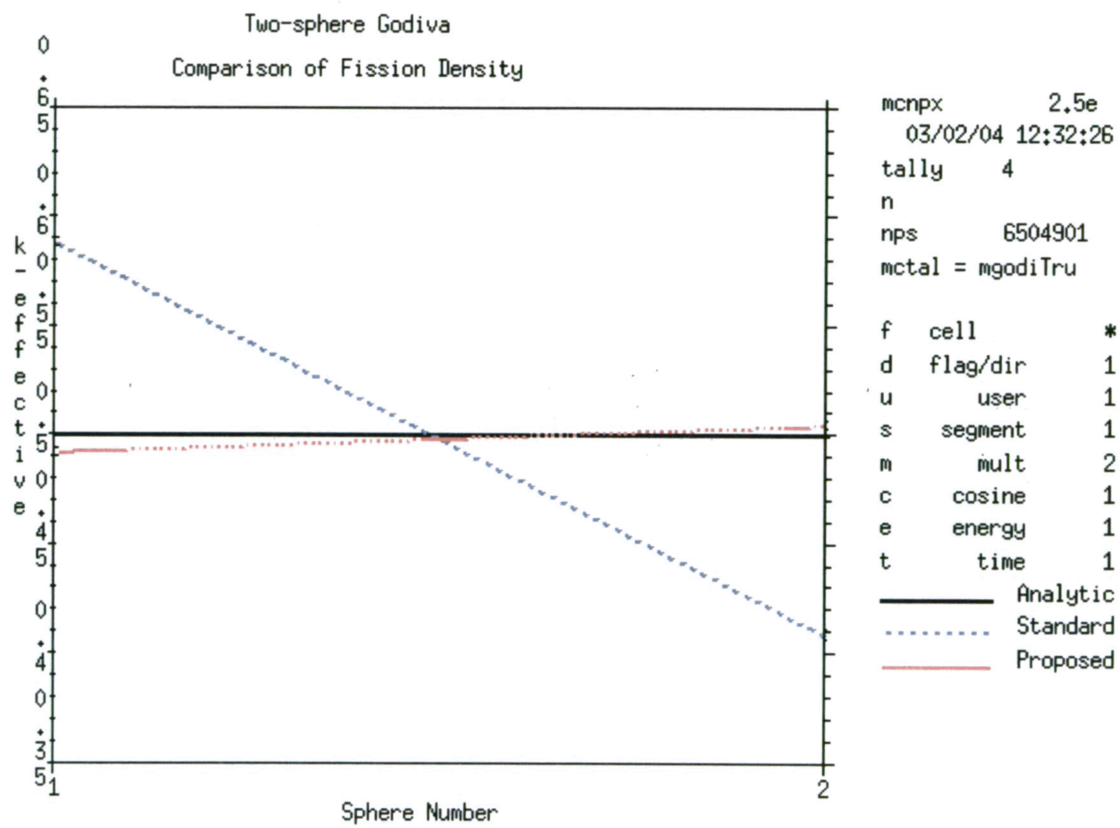


Two-Sphere Godiva

Proposed : 0.1M histories



Two-Sphere Godiva



Fuel Pin Description

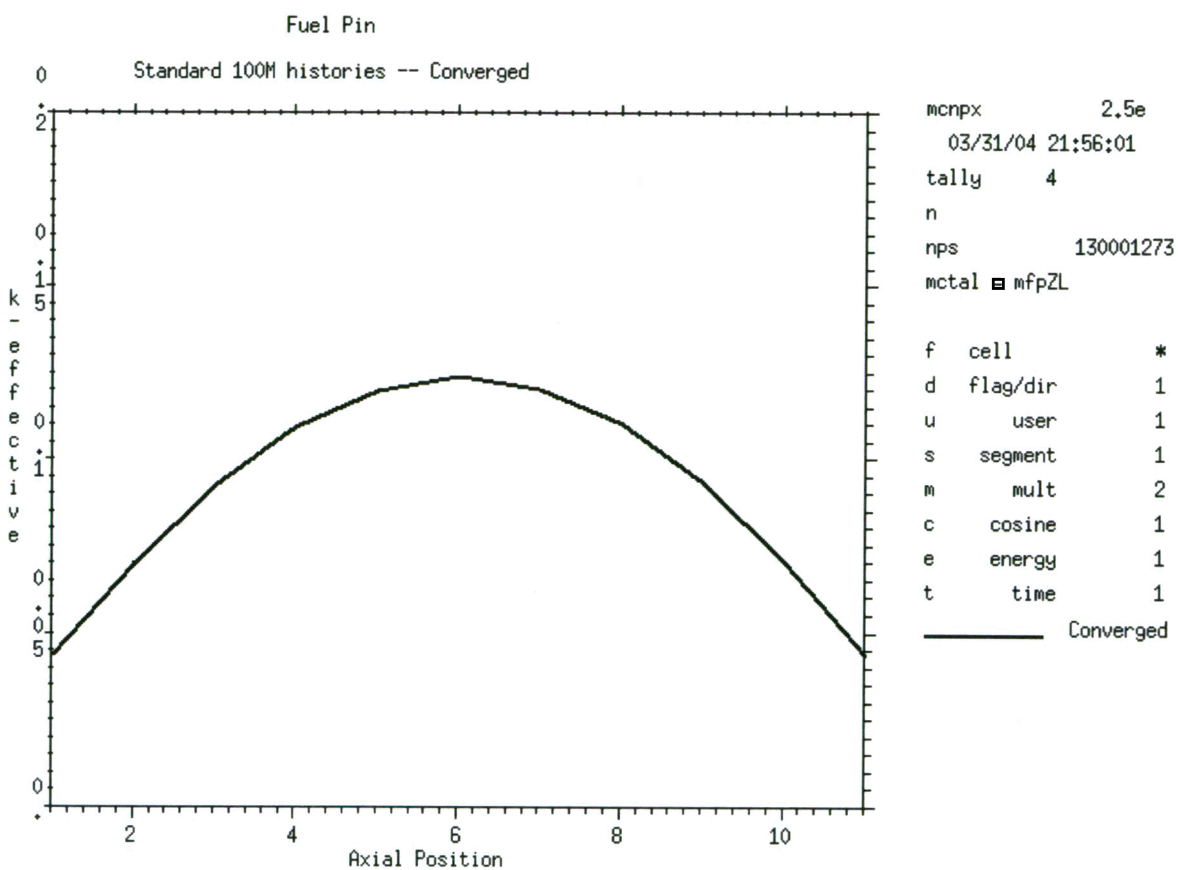
- FUEL
  - Height = 330 cm, Radius = 4 cm
  - 3 wt% ^{235}U and 97 wt% ^{238}U

- CLADDING
  - Height = 330 cm, Thickness = 0.06 cm
  - Zircaloy

- POOL within REFLECTIVE RECTANGULAR SURFACE
  - Height = 334 cm, Sides = 2.926 cm (2 cm gap ea. end)

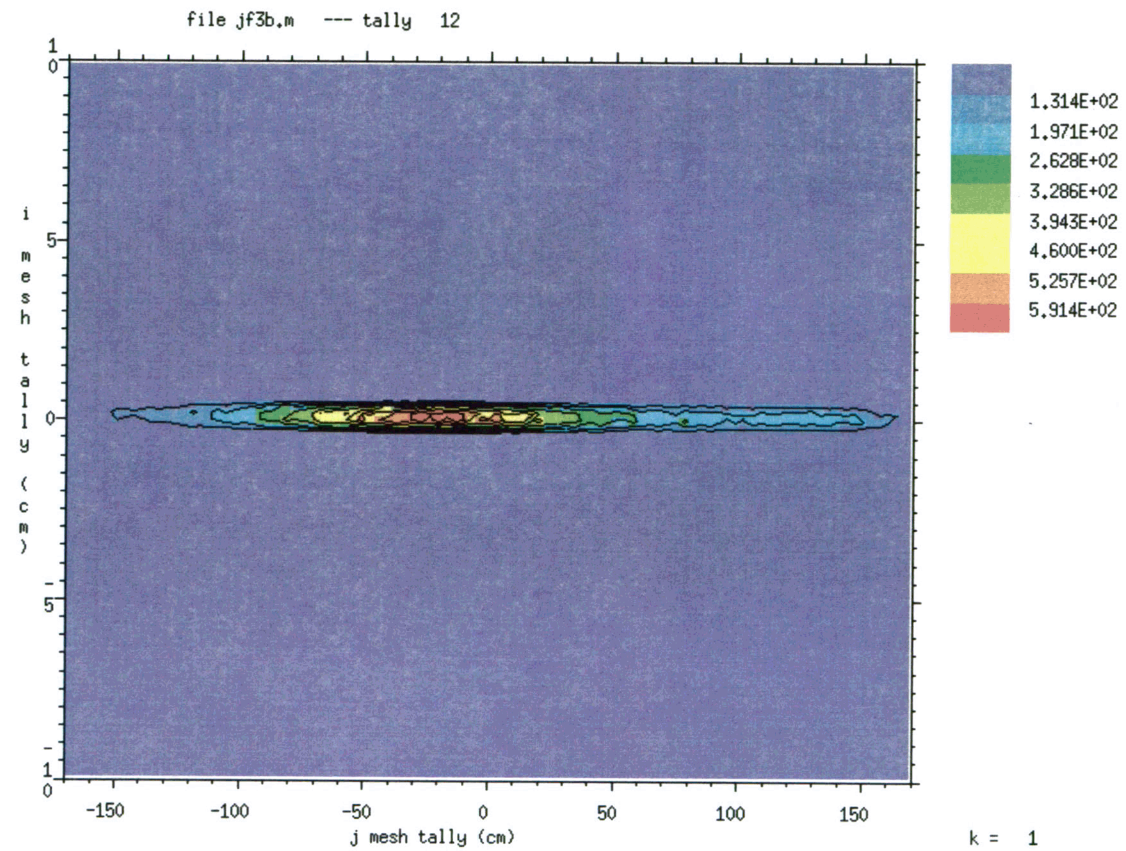
Fuel Pin

Converged : Standard MC : 100M histories



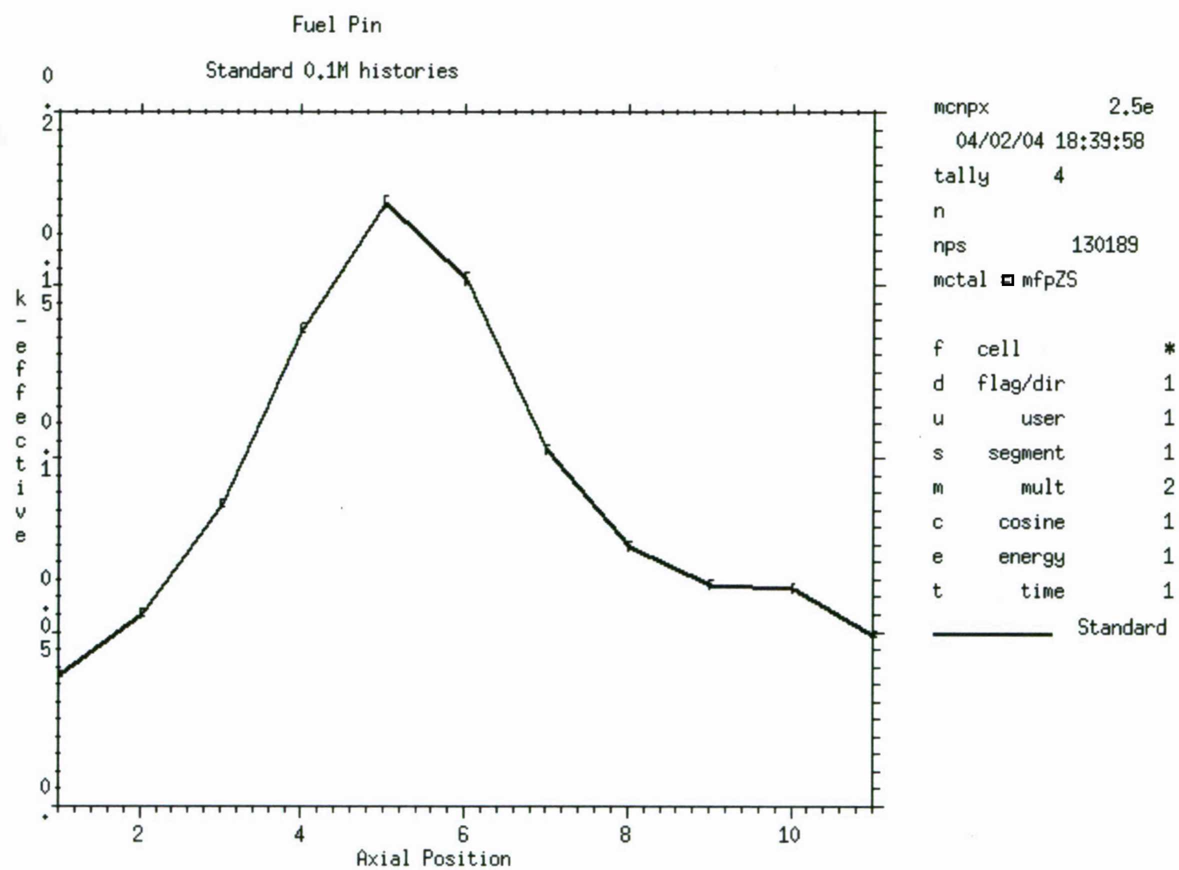
Fuel Pin

Standard MC : 0.1M histories



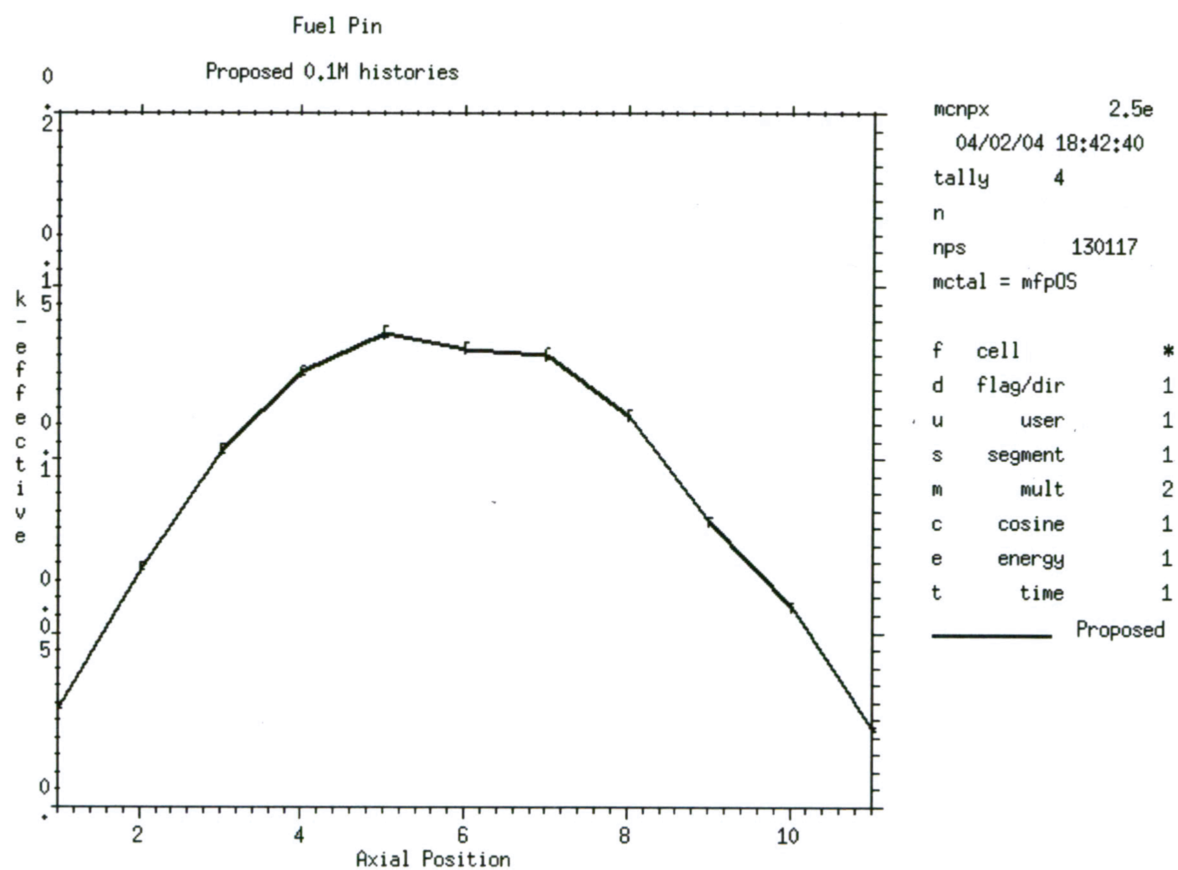
Fuel Pin

Standard MC : 0.1M histories



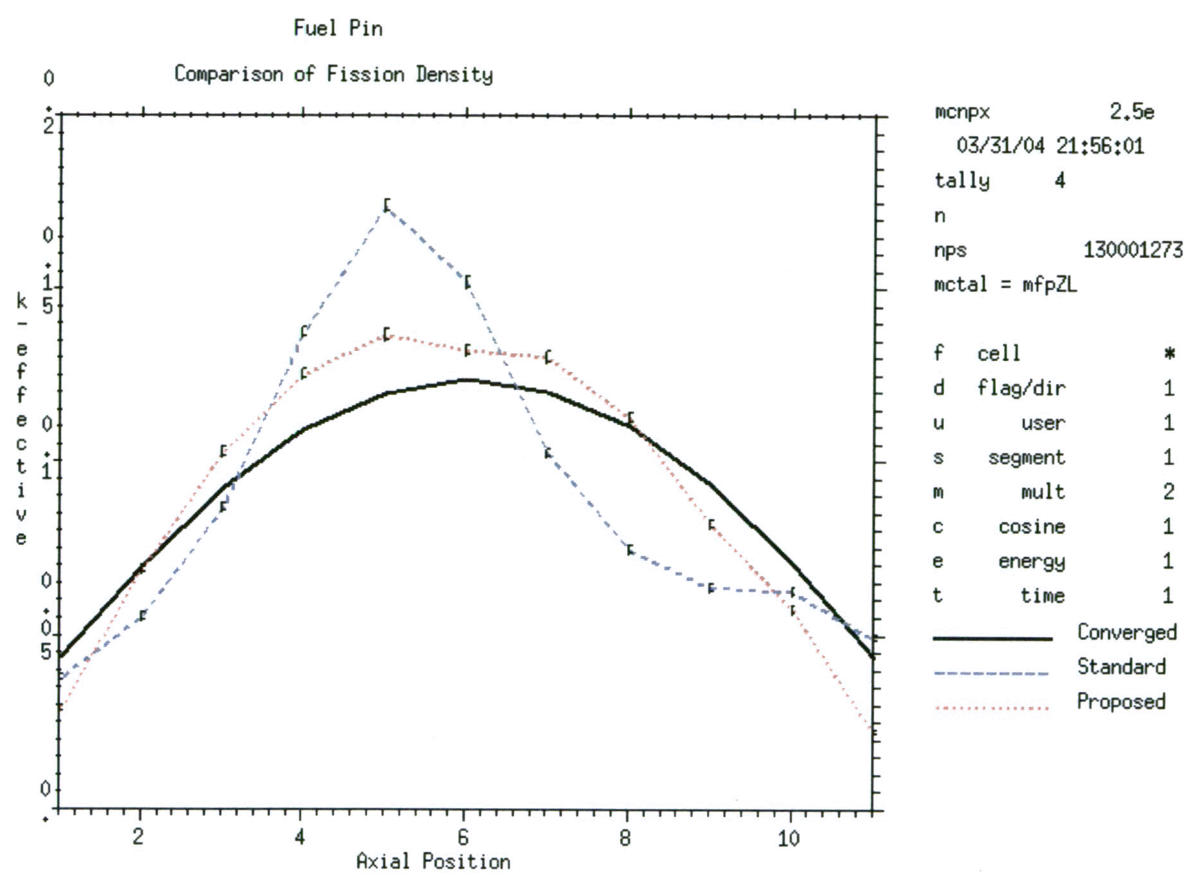
Fuel Pin

Proposed : 0.1M histories



Fuel Pin

Results



Solution Method

- Solve Fission Matrix

$$\mathbf{A}\,\mathbf{x} = k\,\mathbf{x}$$

- Impose Fission Distribution
- Goals:
  - Unbiased
  - Quick
  - Simple
  - Available (MCNPX)

Solution Method

Yamamoto, T., Nakamura, T. and Miyoshi, Y. *Fission source convergence of Monte Carlo criticality calculations in weakly coupled fissile arrays.* Journal of Nuclear Science and Technology **37**(1), 41–52 (2000).

Kitada, T. and Takeda, T. *Effective convergence of fission source distribution in Monte Carlo simulation.* Journal of Nuclear Science and Technology **38**(5), 324–329 (2001).

Kuroishi, T. and Nomura, Y. *Development of fission source acceleration method for slow convergence in criticality analyses by using matrix eigenvector applicable to spent fuel transport cask with axial burnup profile.* Journal of Nuclear Science and Technology **48**(6), 433–440 (2003).

Solution Method

Goals:

- Unbiased
- Quick
- Simple
- Available (MCNPX)

- Find Stable, Approximate Fission Matrix

$$\mathbf{A}\,\mathbf{x} = k\,\mathbf{x}$$

- Nudge Fission Distribution

Method Description

- Construct Fission Source Matrix

Elements of fission matrix, **A**, are P_{ij}

$$P_{ij} = \frac{\sum_{m} W_i \cdot \left(\frac{\nu\sigma_f}{\sigma_t} \right)_j}{N_i}$$

W_i = weight of particle from cell i normalized by the previous cycle k_{eff}

$(\nu\sigma_f/\sigma_t)_j$ = collisional contribution to k_{eff} in cell j

N_i = number of fission sites in cell i

P_{ij} = probability of fission in cell j from a source in cell i per source in cell i

Method Description

- Calculate normalized symmetry matrix (Q_{ij})

$$Q_{ij} = \frac{1}{2} \cdot \left(\frac{P_{ij}}{P_{ii}} + \frac{P_{ji}}{P_{jj}} \right)$$

- Solve **Q** for imposed fission source distribution, **T**

$$\mathbf{Q}\,\mathbf{T} = k\,\mathbf{T}$$

Method Description

- Construct fission production multiplier

$$F_i = T_i / S_i$$

T_i = eigenvector solution via iteration to Q

S_i = fraction of source particles started in cell i = $N_i / \Sigma N_i$

F_i separated into F_i^+ (>1) and F_i^- (<1) components and normalized separately

F_i constrained $0.8 < F_i < 1.2$ for any given cycle

Method Description

- Modify the fission production multiplier such that

$$\Sigma F_i^{m+} S_i + \Sigma F_i^{m-} S_i = 0$$

- Apply multiplier at each collision in the next cycle

$$N_i = (F_i^{m}/k_{eff})(\nu\sigma_f/\sigma_t) + \xi$$

Future Work

- Apply this technique to lattice structures
- Extend the technique to a user-defined mesh
- Investigate approximations of using Q matrix
  - How sensitive is this to asymmetries/inhomogeneities?
- Investigate other solution methods for Q matrix
  - Are there convergence conditions that must be met?
  - Are other methods faster?
- Investigate other normalization techniques

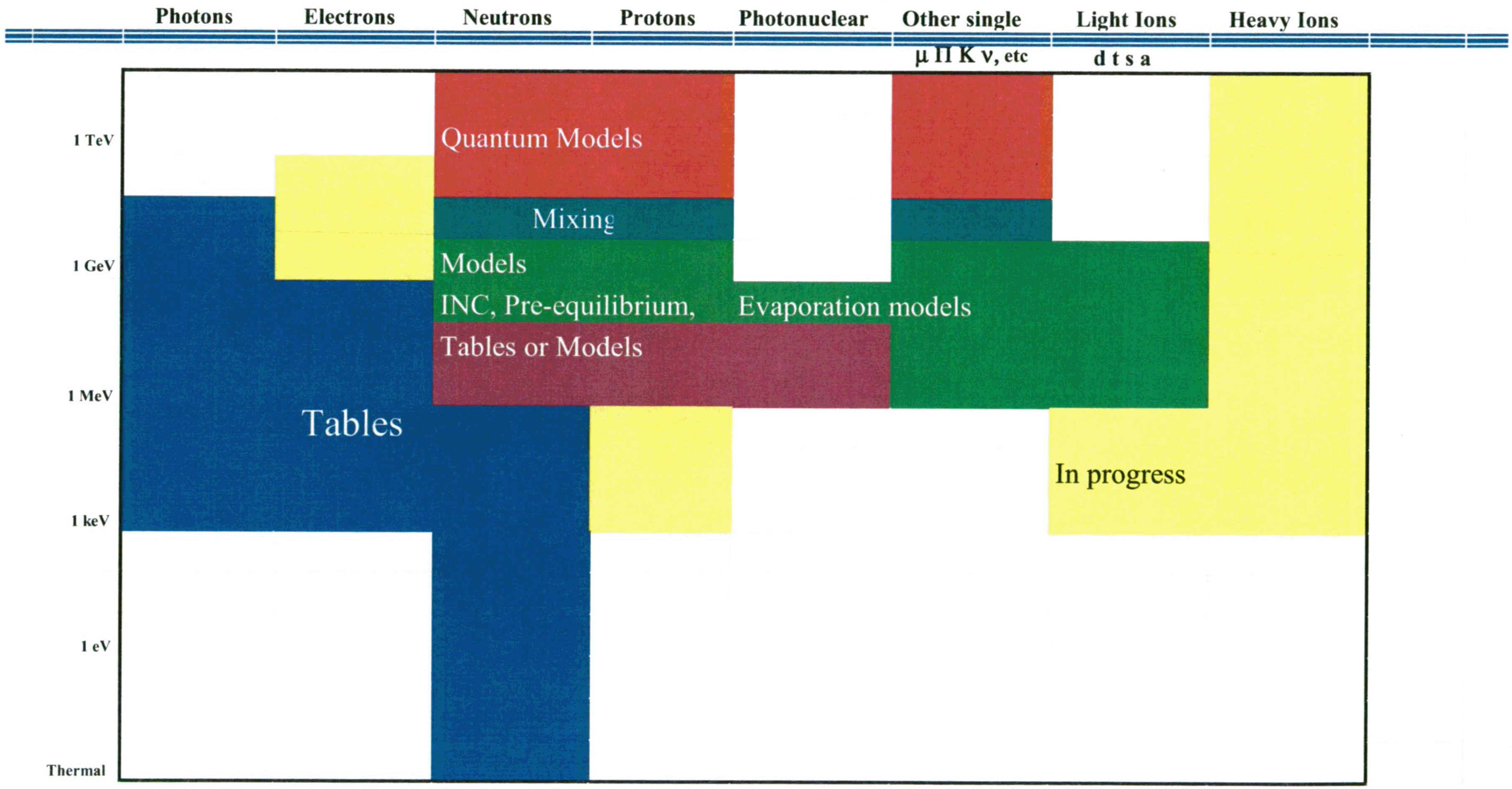
Availability

- Goals:
  - Unbiased
  - Quick
  - Simple
  - Available
- Availability
  - MCNPX 2.5.f
  - 1 subroutine, called 4 times

Why MCNPX ?

- Extended capabilities since 1993
- SSC → APT → ATW → AFCI → GEN IV
- MCNPX extends MCNP4C to 34 particles and much higher energies
- One of 2 modernization approaches for MCNP4C
- Internationalizes MCNP(X)

MCNPX Code Acceptance



MCNPX Applications as of 3/1/04

• Medical	74 groups	212 people
• Fuel Cycles	58 groups	169 people
• Space reactors, cosmics	58 groups	128 people
• Threat Reduction	52 groups	142 people
• Accelerator Driven Systems	46 groups	194 people
• Accelerator Health Physics	37 groups	116 people
• Applied Physics*	37 groups	94 people
• Neutron scattering	16 groups	81 people
• Code development	19 groups	33 people
• Physics models, data eval.	9 groups	18 people
• Nuclear, HE, Astrophysics	9 groups	69 people

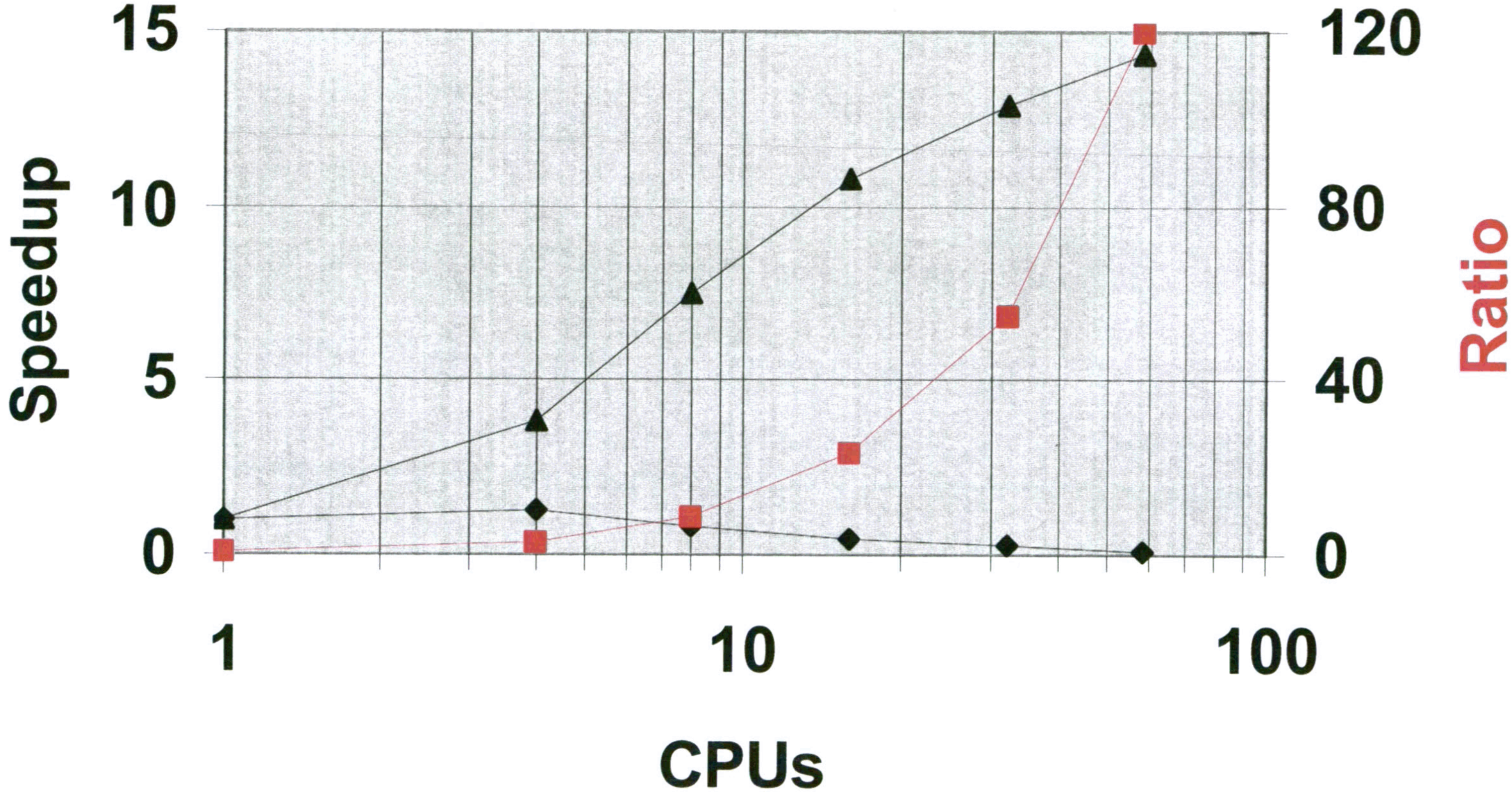
* Radiography, oil well logging, irradiation facilities, isotope production, detector development, environmental, high density energy storage

MCNPX Developers

Los Alamos National Laboratory: John S. Hendricks, Gregg W. McKinney, Laurie S. Waters, Holly R. Trellue, Teresa L. Roberts, Harry W. Egdorf, Eric J. Pitcher, Douglas R. Mayo, Martyn T. Swinhoe, Stephen J. Tobin, Joe W. Durkee, Michael R. James

Oak Ridge National Laboratory:	Franz X. Gallmeier
Purdue University:	Joshua P. Finch, Chan Choi
CEA-Saclay, Gif-sur-Yvette, France:	Jean-Christophe David
Paul Scherrer Institute, Switzerland:	Julian Lebenhaft
HQC Professional Services:	William B. Hamilton

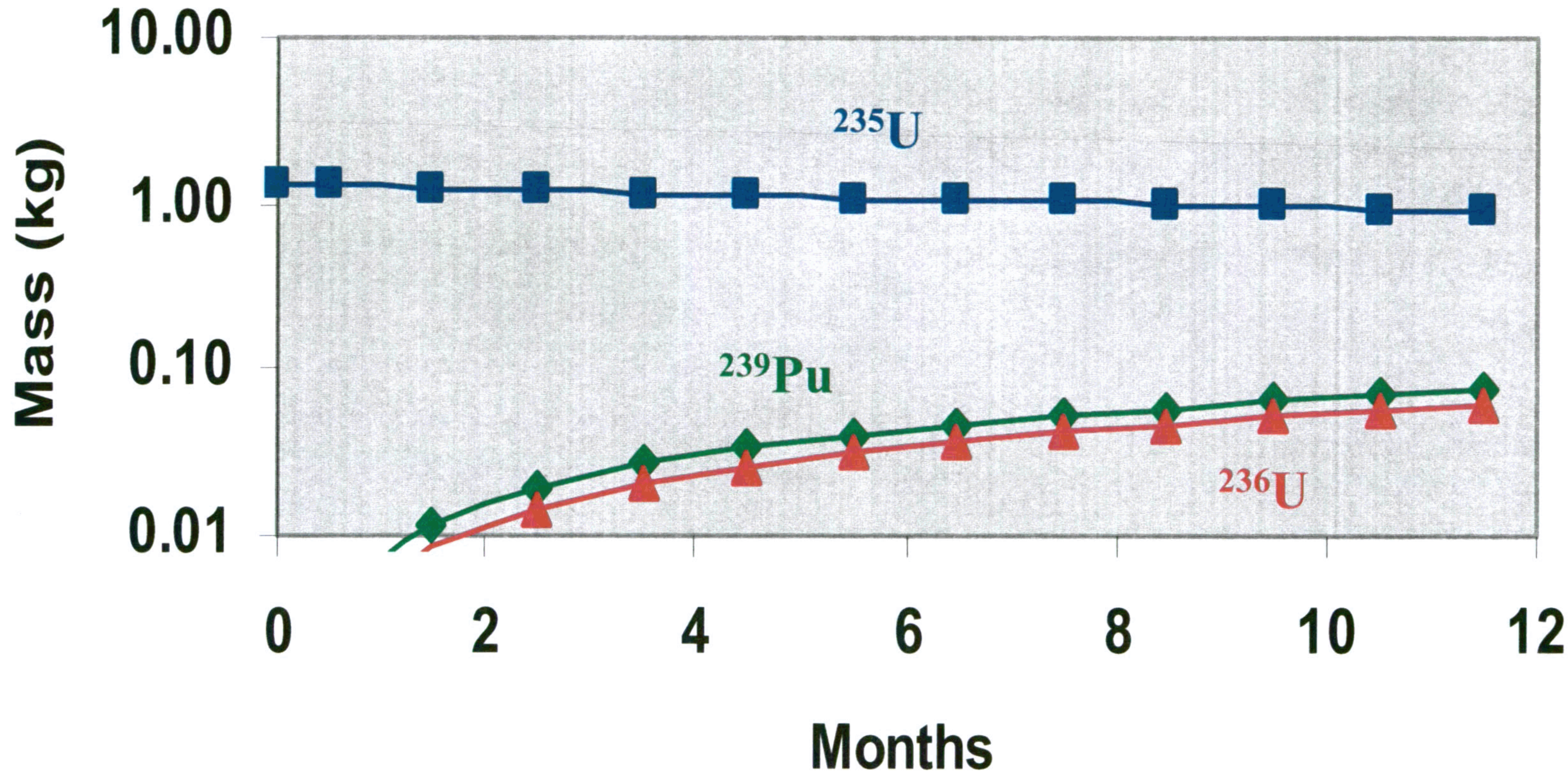
MPI Criticality Speedup



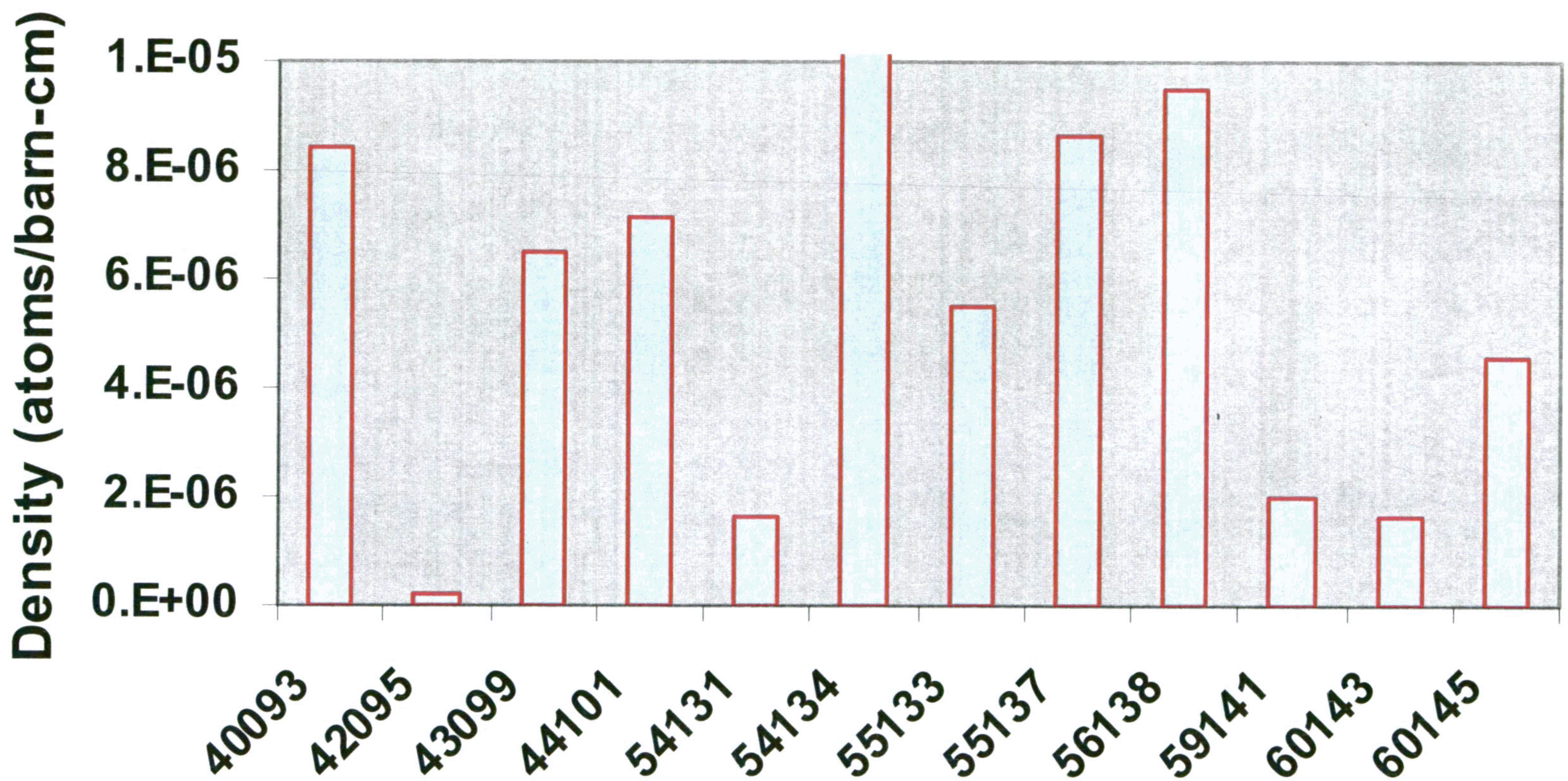
MCNPX Future Capabilities

- Cinder90 capabilities: Burnup / Depletion
- Externally driven sources in near-critical systems
- Mesh tally plot superimposed over geometries
- CAD link
- Pulse-height tallies with variance reduction
- Heavy-ion tracking and interactions
- Magnetic and electric field tracking

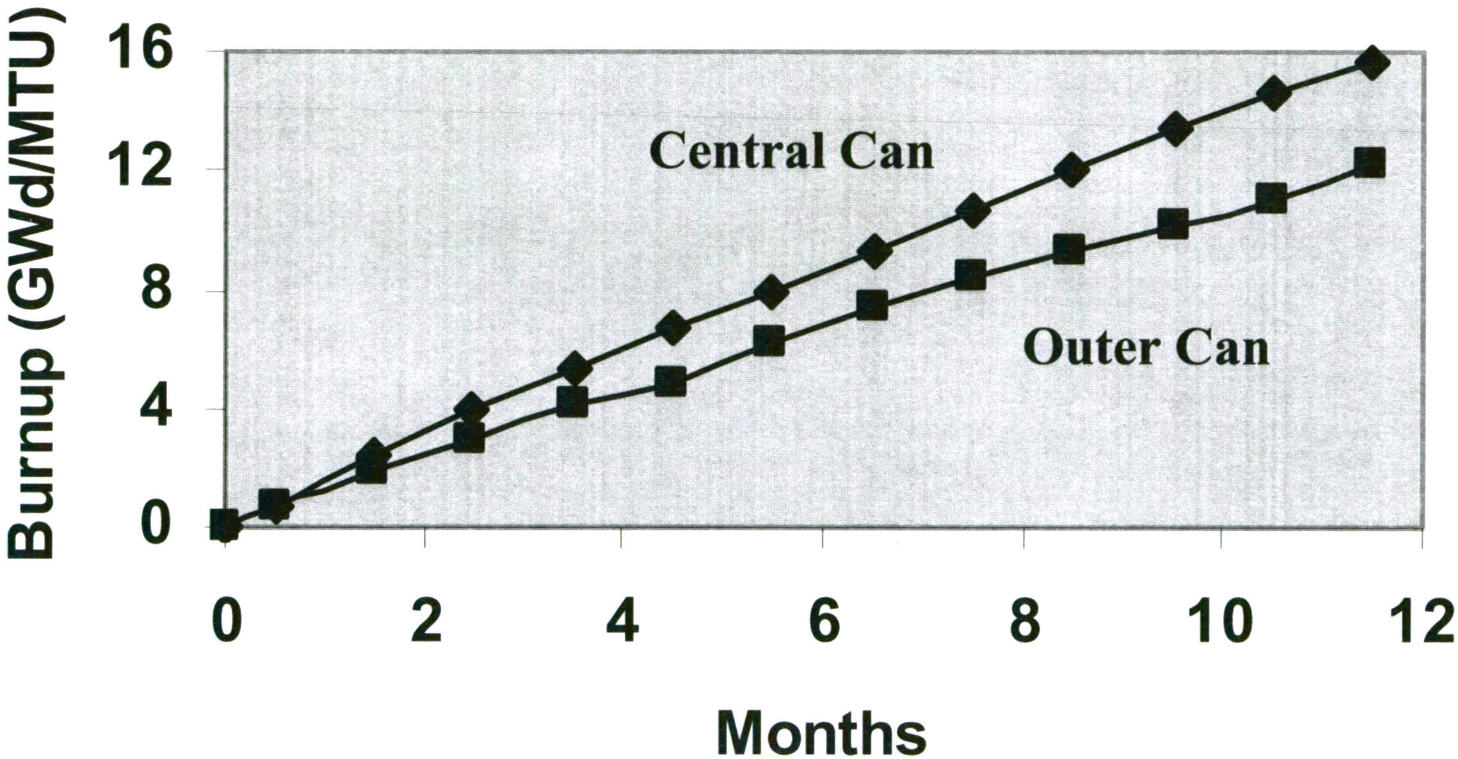
Actinide Inventories



Fission Product Inventories



Can Burnup



Burnup (GWd/MTU)
16
12
8
4
0
Central Can
Outer Can
0
2
4
6
8
10
12
Months